ACCESSION #: 9908300301

NON-PUBLIC?: N

LICENSEE EVENT REPORT (LER)

FACILITY NAME: Nine Mile Point Unit 1 PAGE: 1 OF 8

DOCKET NUMBER: 05000220

TITLE: Reactor Scram Due to Mechanical Pressure Regulator

Suppressor Valve Failure and Mode Switch Position not in

Conformance with Technical Specifications

EVENT DATE: 07/23/99 LER #: 99-004-00 REPORT DATE: 08/23/99

OTHER FACILITIES INVOLVED: N/A DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100%

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:

50.73(a)(2)(i)

50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:

NAME: David F. Topley - Operations Manager TELEPHONE: (315) 349-1752

COMPONENT FAILURE DESCRIPTION:

CAUSE: B SYSTEM: JJ COMPONENT: V MANUFACTURER: C044

B BJ P W315

REPORTABLE EPIX: Y

Y

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On July 23, 1999, at 1558 hours, during post maintenance testing of the turbine mechanical pressure regulator controls, the Nine Mile Point Unit I reactor automatically scrammed from full power from a high neutron flux signal on Channel 11 and Channel 12 of the reactor protection system. Following the reactor scram, operators moved the mode switch through the "Shutdown" position, to the "Refuel" position. The operators did not subsequently reposition the mode switch to "Shutdown" as required by Special Operating Procedure N1-SOP-1, "Reactor Scram." This was a nonconformance with Nine Mile Point Unit 1 Technical Specifications, which permitted the mode switch to be positioned in "Refuel" above 212 degrees F reactor coolant temperature only during reactor coolant system pressure testing, control rod scram timing, and for scram recovery.

Niagara Mohawk Power Corporation determined that the cause of the reactor scram was internal blockage in the pressure suppressor valve in the mechanical pressure regulator feedback loop. The vendor had treated the valve with excessive corrosion inhibitor, and the excessive corrosion inhibitor blocked internal passages. The cause of the inappropriately positioned mode switch was ineffective change management, with contributing factor of inadequate corrective actions for a previously identified issue.

Corrective actions for the reactor scram included placing the plant in a stable condition, cleaning the pressure suppressor valve, and providing guidance for valve disassembly and inspection. Corrective actions for the mispositioned mode switch included revising procedures, and conducting training.

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I. DESCRIPTION OF EVENT

On July 23, 1999, at 1558 hours, the Nine Mile Point Unit 1 reactor

automatically scrammed from a high neutron flux signal on Channel 11 and

Channel 12 of the reactor protection system.

NMPC had previously experienced some pressure oscillations at approximately

85 percent power and suspected some clogging of a mechanical pressure

regulator suppressor valve. Maintenance was undertaken to clean the line

to the valve and replace the valve. The maintenance was completed and post

maintenance testing was started. Operators transferred the turbine

pressure control from the electric pressure regulator to the mechanical pressure regulator (MPR) to complete post maintenance testing. Following the transfer, the Chief Shift Operator (CSO) raised the reactor pressure setpoint in two successive adjustments in order to verify that the MPR controlled reactor pressure. After several subsequent pressure set adjustments, reactor pressure did not respond as expected. Recognizing that reactor pressure was not being appropriately controlled by the MPR, the Assistant Station Shift Supervisor directed operators to perform a manual reactor scram. However, Reactor Protection System Channels 11 and 12 received a high neutron flux signal, which automatically scrammed the reactor before the operators could complete the ordered action. The turbine tripped five seconds after the automatic reactor scram, as designed. Reactor pressure stabilized at 865 psig within 16 seconds after the reactor scram.

No safety relief valves operated during the event. Operators used the bypass opening jack mechanism to open the turbine bypass valves to control reactor pressure, and initiated a controlled cooldown.

Reactor water level during the event reached a low level of +22 inches (8 feet, 10 inches above the top of active fuel), and a high level of +98 inches. The high pressure coolant injection system initiated as designed, starting Feedwater Pump 12 (Feedwater Pump 11 was already operating). Feedwater Pumps 11, 12, and 13 were providing water to recover the reactor inventory. During the transient, Feedwater Pumps 11 and 12 tripped on low suction pressure because of excessive feedwater flow, predominately from high flow on Feedwater Pump 13. Feedwater Flow Control Valve 13 responded more slowly than expected to a close signal as part of a recently installed setpoint setdown design change. Feedwater Pump 13 is geared to the turbine and supplies water during turbine coastdown. Operators recognized the feedwater pump trips and restarted Feedwater Pumps 11 and 12, and reset the high pressure coolant injection system signal. Subsequently, in accordance with the emergency operating procedures and Special Operating Procedure NI-SOP-1, operators tripped Feedwater Pumps 11 and 12 and Control Rod Drive Pump 11 prior to level reaching +95 inches. Reactor level rose to +98 inches, and reactor water entered the bottom of the emergency condenser steam lines (the bottom of the emergency condenser steam supply nozzles are at +97 inches). Water level control was re-established with the reactor water cleanup system, and within two-and-one-half minutes, the water level returned to below the emergency condenser steam supply nozzles. The emergency condensers were not required to operate.

I. DESCRIPTION OF EVENT (Cont'd)

Following the reactor scram, operators moved the mode switch through the "Shutdown" position and placed the mode switch in the "Refuel" position in accordance with Special Operating Procedure N1-SOP-1, "Reactor Scram." Special Operating Procedure N1-SOP-1 directs concurrent operations, including placing the mode switch back to "Shutdown" after operators verify that all control rods are fully inserted. The day shift Station Shift

Supervisor did not complete the action to place the mode switch to "Shutdown" because of prioritization with other actions to be completed, and his understanding that the "Refuel" position was acceptable for Scram Recovery. The oncoming night shift Station Shift Supervisor received turnover status that the mode switch was in "Refuel" but he also believed that it was acceptable since the plant was in Scram Recovery and the switch change could be accomplished later. The crew did not re-evaluate the completion of this step until after the reactor coolant system was cooled to below 212 degrees F. The failure to reposition the mode switch in "Shutdown" shortly after verifying that the control rods had been inserted, and while the reactor was still greater than or equal to 212 degrees F, was a nonconformance with Technical Specification Definition 1.1.b, "Shutdown Condition - Hot."

II. CAUSE OF EVENT

Niagara Mohawk Power Corporation (NMPC) conducted a root cause investigation and determined that the cause of the reactor scram was the failure of the MPR pressure suppressor valve (a J. A. Campbell Company Micro-Bean Valve). The vendor treated the valve with excessive corrosion inhibitor. NMPC was not made aware that the vendor had treated the valve with corrosion inhibitor during valve assembly. During the review of this event, NMPC tested the valve and verified that the excessive corrosion inhibitor prevented satisfactory feedback to the MPR, thereby causing the loss of pressure control while performing the setpoint changes.

NMPC determined that the cause of the inappropriate positioning of the reactor mode switch was ineffective change management. In 1988, Technical Specification Amendment 99 added a footnote to Technical Specification Definition 1.1.b, indicating that the mode switch could be placed in the "Refuel" position at reactor coolant temperatures above 212 degrees F in order to perform reactor coolant system pressure testing, control rod scram timing, and scram recovery operations. The NRC Safety Evaluation Report for Amendment 99 indicated that the condition of scram recovery operations applied to manually inserting control rods that had not fully inserted.

NMPC did not adequately implement Technical Specification Amendment 99 in its procedures or training to ensure that Refuel position limitations were controlled and understood.

II. CAUSE OF EVENT (Cont'd)

A contributing cause was that NMPC took inadequate corrective actions for a previously identified issue. Nine Mile Point Unit 1 Licensee Event Report 97-11 reported that previous normal plant shutdowns were not in conformance with Technical Specifications. The corrective actions from that report included changing procedures and training operators. In evaluating the issues in LER 97-11, NMPC did not recognize the underlying knowledge error in the understanding of "scram recovery." This was a missed opportunity to identify and correct the problem prior to this event.

NMPC determined that the cause of the low suction pressure trip of

Feedwater Pumps 11 and 12 was a combination of a potentially inherent design issue and an inappropriate controller setup involving the feedwater Flow Control Valve 13 controller setpoint setdown modification that was installed during Refuel Outage 15. Previous operating experience has demonstrated that a feedwater pump trip on low suction pressure occurs following some, but not all, scrams. The hydraulics of the feedwater system were not specifically evaluated previously. The modification reset the controller setpoint from controlling at +72 inches nominally, to controlling at +45 inches following a scram. NMPC installed the modification to prevent overfilling the reactor vessel, such that with the proper feedwater flow control valve operation, reactor level should not exceed +90 inches. During the event, the controller's output signal saturated and Flow Control Valve 13 responded more slowly than expected. The large flow through Flow Control Valve 13 contributed to the low suction pressure trip of Feedwater Pumps 11 and 12.

III. ANALYSIS OF EVENT

NMPC is reporting the reactor scram in accordance with 10CFR50.73(a)(2)(iv); "Any event or condition that resulted in a manual or automatic actuation of any engineered safety feature (ESF), including the reactor protection system (RPS)..."

NMPC is reporting the inappropriately positioned mode switch in accordance with 10CFR50.73(a)(2)(i)(B); "Any operation or condition prohibited by the plant's Technical Specifications..."

The reactor scram signal was the reactor protection system design response to a high neutron flux condition. All control rods fully inserted in response to the reactor scram signal. Operators reviewed scram times for selected control rods and concluded that the insertion times for these control rods met acceptance criteria. The plant

III. ANALYSIS OF EVENT (Cont'd)

response was consistent with the Updated Final Safety Analysis Report description of a reactor automatic scram, followed by a turbine trip five seconds later.

The high pressure coolant injection system at Nine Mile Point Unit 1 is neither an engineered safety feature nor credited in the Appendix K loss of coolant accident analysis. The system is not safety-related. There is no specific analytical time limit imposed on high pressure core injection system initiation for any accident analysis. NMPC engineering performed an Engineering Supporting Analysis in regard to the Feedwater Pump 11 and 12 trips. The analysis reviewed the requirements for feedwater flow, and concluded that feedwater pumps operating in the high pressure coolant injection mode might trip following a reactor scram. However, the tripping of Feedwater Pumps 11 and 12 on low suction pressure does not result in the failure to maintain feedwater flow. Turbine driven Feedwater Pump 13 provides minimum high pressure coolant injection flow for approximately 3.2 minutes during coastdown of the turbine, independent of the other two feedwater pumps. During this event, operators restarted the pumps within twenty seconds. Past experience has proven that the operator training and the proceduralized actions of Procedures NI-SOP-1, NI-OP-16 (Feedwater System Booster Pump to Reactor), and N1-ARP-H3 (Control Room Panel 113) are effective in restarting the tripped pumps. The temporary compensatory measure of crediting operator action to restart the tripped feedwater pumps complies with the corrective action guidance of NRC Generic Letter 91-18 (Resolution of Degraded and Nonconforming Conditions and Operability Determinations) and NRC Information Notice 97-78, "Crediting of Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times."

The emergency condensers were not required to operate during this event. Following the event, NMPC engineering evaluated the introduction of reactor water into the emergency condenser lines and determined that there were no adverse effects. In this transient, there was no demand for the emergency condensers to initiate. Additionally, in order to protect the piping from hydraulic transients caused by water in the piping, NMPC had installed low point drains, and hydraulic snubbers at piping direction changes. The emergency condenser steam lines are also sloped toward the reactor. Engineers performed a detailed visual inspection and concluded that there was no evidence of damage to the emergency condenser piping, pipe supports, or insulation as a result of this event.

Additionally, Engineering Services personnel performed a probabilistic risk assessment, which concluded that the conditional core damage probability for this event was 2.8E-06. The analysis was conservative in that the main condenser was not lost, the emergency condensers were not required to operate, and Feedwater Pumps 11 and 12 were immediately restarted. NRC letter SECY-98-298, "Status Report on Accident Sequence Precursor Program and Related Initiatives," stated that the accident Sequence Precursor Program considered precursors with conditional core damage probabilities of greater than or equal to 1.0E-04 to be important with respect to risk significance. Based on this analysis, this event was not important with respect to risk significance.

III. ANALYSIS OF EVENT (Cont'd)

There was no adverse effect associated with the inappropriately positioned mode switch. Interlocks associated with the "Refuel" position ensure that only one control rod may be manipulated at a time. The reactor core loading requirements of Technical Specification 3.1.1.a, "Reactivity Limitations," ensure that the reactor will remain subcritical at any time in the operating cycle with the most reactive control rod fully withdrawn and all other operable control rods fully inserted. Engineering Services personnel performed a probabilistic risk assessment and concluded that the failure of the operators to return the mode switch to the "Shutdown" position was not risk significant.

Based on the above analyses, these events posed no threat to the health and safety of the general public or plant personnel.

IV. CORRECTIVE ACTIONS

1. NMPC cleaned, reinstalled, and tested the J. A. Campbell Company Micro-Bean Valve.

2. NMPC added notes to the electronic Work Order to ensure that future J. A. Campbell Company Micro-Bean Valves are disassembled and inspected prior to installing the valves.

3. Special Operating Procedure NI-SOP-I and all appropriate operating procedures were revised to require placing the reactor mode switch in "Shutdown" immediately following any reactor scram. On shift training of each operating crew will be completed as the crews come on shift. This activity will be completed by August 30, 1999.

4. NMPC will review selected Technical Specifications Amendments to verify adequacy of the implementation by October 31, 1999.

5. NMPC will conduct classroom and simulator training concerning the inappropriate positioning of the mode switch and expectations for exiting Special Operating Procedures for operations personnel by
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October 22, 1999.

6. Generation branch managers will ensure that department procedure preparers and reviewers understand their responsibility to ensure that sufficient research is conducted to understand the basis for actions being changed in procedures by November 25, 1999.

IV. CORRECTIVE ACTIONS (Cont'd)

7. NMPC completed a design change to the Flow Control Valve 13 flow controller to limit the output signal at 50 milliamps DC to preclude saturation of the output signal.

8. NMPC will evaluate the feedwater system hydraulics associated with this event and previous scrams to determine applicable failure modes associated with the feedwater pump trip and determine whether further modifications may be required by October 28, 1999.

9. NMPC will develop preventive actions to address the inappropriate controller setting by October 28, 1999.

V. ADDITIONAL INFORMATION

A. Failed components:

o Turbine controls MPR pressure suppressor valve (J. A. Campbell Company Micro-Bean Valve)

o Feedwater Pumps 11 and 12

B. Previous Similar events:

Licensee Event Report 97-11, "Previous Plant Shutdown in Violation of Technical Specifications," reported similar occurrences of inappropriately placing the mode switch in the "Refuel" position during plant shutdowns. Corrective actions for that event included changing operations procedures, training operators, and reviewing procedures. The corrective actions from that event could have prevented the inappropriate positioning of the mode switch reported in this Licensee Event Report.

V. ADDITIONAL INFORMATION

C. Identification of components referred to in this Licensee Event

Report:

Components IEEE 803A Function IEEE 805 System ID

MPR RG JJ

Electric Pressure Regulator RG JJ

Pressure Suppressor Valve V JJ

Turbine Bypass Valves V JI

Main Turbine TRB TA

Safety Relief Valves RV N/A

Bypass Opening Jack Mechanism N/A TA

High Pressure Coolant Injection

System N/A BJ

Feedwater Pumps P BJ

Control Rod Drive Pump P AA

Emergency Condensers COND BL

Reactor Water Cleanup System N/A CE

Hydraulic Snubbers SNB BL

Control Rods N/A AA

Emergency Condenser Steam

Supply Nozzles NZL BL

Reactor Mode Switch 33 AA

Reactor Protection System N/A JC

Piping Supports SPT BL

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Niagara Mohawk [Registered Trademark]

August 23, 1999

NMP1L 1456

U. S. Nuclear Regulatory Commission

Attn: Document Control Desk

Washington, DC 20555

RE: Docket No. 50-220

LER 99-04

Gentlemen:

In accordance with 10 CFR 50.73(a)(2)(i)(B) and 10 CFR 50.73(a)(2)(iv), we are submitting LER 99-04, "Reactor Scram Due to Mechanical Pressure Regulator Suppressor Valve Failure and Mode Switch Position not in Conformance with Technical Specifications."

Sincerely,

Robert G. Smith

Plant Manager - NMP1

RGS/CES/kap

xc: Mr. H. J. Miller, NRC Regional Administrator

Mr. G. K. Hunegs, Senior Resident Inspector

Records Management

Nine Mile Point Nuclear Station, PO. Box 63, Lycoming, New York 13093-0063

o 315.343.21 10 o www.nimo.com

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